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AN APPLICATION OF SYSTEMS ANALYSIS TECHNIQUES TO THE STUDY OF REACTOR SEISMIC SAFETY *

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ABSTRACT

Systems analysis is being used in conjunction with structural analysis to study the conservatisms and to provide insights into aspects of reactor seismic safety. An event-tree/fault-tree model of a commercial nuclear power plant is being constructed to determine the probability of release and probabilities of system and component failures caused by possible seismic events. The event-tree/fault-tree model is evaluated using failure data generated by applying the response a component sees to the component's fragility function. The responses are calculated by a structural analysis code using earthquake time histories as forcing functions. The quantification of the event-tree/fault-tree model is done conditional on a given seismic event and the conditional probabilities thus calculated unconditioned by integrating the results over the seismic hazard curve. In this way, most of the dependencies between event failures resulting from the seismic event itself are removed making known fault-tree analysis quantification techniques applicable. The outputs from the computations will be used in sensitivity studies to determine the key calculations and variables involved in seismic analyses of nuclear power plants.

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1. INTRODUCTION

Lawrence Livermore Laboratory is currently conducting a large multi-year seismic research program for the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, entitled the Seismic Safety Margins Research Program (SSMRP) as described by Smith, et al., [1]. One of the purposes of this program is to provide estimates of the conservatism in current NRC seismic safety requirements and to develop improved requirements. The program is broken into three phases. The initial phase centers on development of a probabilistic methodology that will more realistically estimate the behavior of nuclear power plant structures and systems during hypothesized earthquakes including a characterization of the uncertainties. This initial phase will yield estimations of the probability of failure of structures, systems and components and the probability of radioactive releases over a range of earthquake levels. These estimates will then be used to establish priorities for future research and analysis to be carried out in phase 2. In phase 3 recommendations for changes in NRC safety requirements will be made by developing a balanced, deterministic, seismic design methodology. The SSMRP can be thought of as an application of risk methods to validate or improve licensing processes as they concern seismic design.

A number of seismic safety studies have been made of nuclear power plants. Typical of these are those by Ang and Newmark and Cornell and Newmark [2,3]. In most cases these studies were based on failures of specific critical components and/or subsystems. One study which analyzed the seismic safety of a complete nuclear power plant, Diablo Canyon, [4] used an event-tree/fault-tree representation of the reactor systems to predict risk to the public due to seismic events. The event-trees/fault-tree methodology used was based to a large extent on WASH-1400 [5] results.

In all these seismic safety studies three elements are involved in the analysis. The first element is a seismic hazard curve which plots the probability of exceeding an earthquake of a given intensity against that intensity factor. The intensity factor commonly used is the peak ground acceleration. Another element of these analysis is some sort of characterization of the components or structures resistance to failure. This may be a plot of the probability of failure versus the same earthquake intensity factor. The third aspect of the analysis is the systems model which may be a simple block diagram or a complex event-tree/fault-tree model, depending on the complexity of the system and whether it is a specific or generic study. The idea of the analysis is then to use the resistance characterization and the system model to determine the probability of system failure given a certain intensity earthquake. This probability of failure is then convolved with the seismic hazard curve to come up with an unconditional probability of failure or in our case probabability of release.

SSMRP, in its initial phase, is not so much a program to determine the risk due to seismic events at nuclear power plants but instead is using risk assessment techniques to prioritize research and estimate conservatisms inherent in current design methods. Probabilistic methods will be used and uncertainties propagated throughout the calculations so that point estimates alone will not have to be relied on. Output results will be characterized by confidence limits or some other measure of the upper and lower bounds of the analysis. Those parts of the calculations and those input variables that contribute the most uncertainties in the results will be prime candidates for future research.

2. COMPUTATIONAL PROCEDURE

The computationl procedure to be described can be thought of in two parts. In the first part (response calculation), the seismic input information in the form of time-histories, is used as a forcing function for calculation of the response of the structures and components. The second part of the calculation (systems calculation) will involve the quantification of the event/fault-tree model of the reactor systems. This calculational chain is illustrated in Fig. 1.

The first block in the response calculation, seismic input, processes the regional seismic characteristics, source parameters, site modification factors and other seismic related input and generates as output a characterization of the free-field ground motion. The free-field motion is then used as an input into the second block, soil-structure interaction. Together with the underlying soil data and reactor structure information a basemat motion is calculated or, alternatively, the soil and the structure are modeled together to determine the structural response. The major structural response information is then used as input either to subsystem analysis or directly into the second part of the code where the component failure analysis is carried out. The response data may be in the form of peak accelerations, velocities, displacements, stresses, strains, etc.

Each calculation is carried out for a set of time histories with a common peak ground acceleration (a_p) and spectral shape parameter (f_c). The responses calculated from this time history set are then averaged with the various other input quantities such as soil shear modulus, damping, or structural stiffness Calculation of average responses, standard deviations, and correlation coefficients at several hundred component locations may be involved. This response data is stored in vector form as shown in Fig. 2. Attributes can be incorporated in the response vector for later sensitivity studies.

The response vector is then input, along with fragility functions for all the structures and components for which failure must be calculated, into the systems analysis part of the calculation [6]. An example of a fragility function is shown in Fig. 3. Plotted on the ordinate is the probability of failure and on the abcissa is the response quantity which will be used to determine the probability of failure e.g., peak acceleration, peak stress. Random uncertainty is characterized by the slope of the median value for the failure probability (solid curve). Uncertainty in this median value is depicted by upper and lower bounds as shown by the dotted lines. The range between the dotted lines may be thought of as modeling or systematic uncertainty as opposed to random uncertainty. Both types of uncertainty will be studied in the SSMRP.

Table 1 lists the seven computation options available in the systems calculation. These reflect choices between Monte Carlo or analytical computation of failure and/or binary or probabilistic evaluation of the event and fault trees. In the analytical computation of failure covariance matrices and mean vectors of responses and resistances are combined to determine the measured standard deviations of the output. This technique allows for only normal or log-normal distributions of the input variables but joint failure probabilities can be considered. The Monte Carlo technique uses correlated random numbers generated using the covariance matrices. Any distribution of input variables that can be characterized by two moments can be used in the Monte Carlo type of computation.

Another choice is available when determining how to evaluate the event and fault trees. If the component failure probabilities are high, as would be the case with very large earthquakes, then a binary characterization of failure would lead to more efficient computations. In the binary approach random numbers are applied to failure probability distributions to decide whether the components are in a failed or non-failed state and the event and fault trees evaluated accordingly. A unique accident sequence is generated for each computation run. After many runs the talley on each accident sequences gives the overall probability that that accident sequence would occur. In the probabilistic calculation, failure probabilities are assigned each component and the probabilities are multiplied in the event trees to get the probability of all the accident sequences. This is more efficient if the failure probabilities of the components and systems are less than 10^{-2}. The overall strategy of the systems calculation is shown in Fig. 4.

3. SENSITIVITY ANALYSIS

Sensitivity analysis is the identification of important variables, the ordering of the variables according to importance and an indication of the significance of the ordering. For SSMRP significant input variables may be peak free-field ground acceleration, structural damping, structural stiffness, shear modulus of the soil, etc. Also of importance may be modeling variables such as the coefficients used to describe the motion of the seismic waves from the source to the ground surface or the order of the polynomial used to approximate a functional relationship. Besides relating the variables or prioritizing them with respect to each other it is important to relate them to noise.

In SSMRP we will use a number of methods to do our sensitivity studies. Our ultimate goal is to come up with models of the outputs which can be used not only to give a measure of the relative importance of the various input variables but will be useable as a response surface. Such a response surface could be used with Monte Carlo simulation techniques for further studies [7].

Other methods to determine sensitivity would be to construct partial derivatives of the output with respect to the input variables. This can be done with respect to reference values of the input variables or the derivatives can be averaged over the range of the input variables. We will also be doing dominance analysis to identify important accident sequences, the most important safety systems and the key components in the safety systems. This ranking can be done for specific earthquakes or for a range of earthquakes exceeding a given magnitude.

4. EVENT-TREE/FAULT-TREE DEVELOPMENT

In SSMRP we wish to have as complete a measure of the consequences of seismically induced failure as possible. For this reason we are using a event-tree/fault-tree representation of an existing nuclear power plant (Zion I) in order to conduct our studies.

Event trees are used to identify important accident sequences which can lead to radioactive releases. For the SSMRP, we are initially only considering sequences which can lead to core melt, since most of the radioactive materials inside the fuel element cannot be released until the core melts. However, the technique itself is not limited to consideration of just core melt accidents.

An event tree is constructed for each of several initiating events, e.g., loss-of-coolant-accidents of various sizes and reactor transients caused by such occurrences as loss of offsite power. The outcomes of such initiating events are determined by the operations of systems which have an effect on the event. Figure 5 shows an event tree whose initiating event is a pipe break leading to a large loss-of-coolant-accident [8]. A number of systems have an effect on the outcome of this accident and are designated by the letters across the top of the figure. These systems include parts of the emergency core cooling system and fission product removal systems. A containment failure event tree (not shown) is attached to the end of each accident sequence and used to determine the type of release from a given accident sequence. Successful operation or failure of each system determines the accident sequence and the multiplications of system failure probabilities determines the accident sequence probabilities.

The failure probabilities for the systems included in the event trees are determined by constructing fault trees for these systems. Failure probabilities as determined from the response calculations and the fragility functions are assigned to the basic events in the fault trees and the top event probability calculated from the Boulean representation of the tree. Figure 6 pictorially shows how the event trees and fault trees are used in SSMRP.

Problems concerning the statistical dependence between events and data insufficiencies are more acute than for a random failure analysis like WASH-1400. By constructing the event and fault trees conditional on the seismic event, some of the concern about dependencies is removed. Where identified, dependencies are being modeled in the trees explicitly. During the quantification process the trees will be evaluated for a given seismic event. The conditional probabilities so calculated will be unconditioned by convolving with the seismic hazard curve. In this way, most of the dependency resulting from the common cause nature of the seismic event is removed.

Another source of dependency results from the commonality of components. The fragility function used to calculate the probability of failure of the component may be for a population of components from different manufacturers. If the components in a given plant are from the same manufacturer there is a correlation between the fragilities for these components. The computational procedure will be able to handle correlation between these fragilities in one of its calculational modes. Also, bounding techniques will be used to bracket the dependency problem and more detailed common cause analysis will be used in critical areas and where data permits.

In order to supplement the meager amount of test data to failure available on the components in a nuclear power plant, expert opinion and panel review will be used. Sensitivity studies will also be carried out on expert opinion to see how sensitive the final results are to expert opinion and in what areas. By soliciting expert opinion and combining it where possible with test data and reviewing the results with a knowledgeable panel of experts it is hoped that as good a data base as possible can be constructed. Although we may strive for consensus opinion in some cases, care will be taken to make sure that the opinion of the individual experts represent an independent data set.

5. CONCLUSION

The SSMRP represents a step forward in the application of systems analysis techniques to structural reliability problems. In order to make the problem tractable, certain simplifications have to be made. These simplifications will only be made in areas where it is clear that important matters will not be obscured. Some of the structural analysis will be done in a very sophisticated manner and used to verify the more simplified models that are used. Fault trees are used to model system failure but are not going to be relied upon to model the failure of the structure, structures being highly redundant. Rather the fault trees will reflect judgments on structural failure modes provided by the structural analyst. Fault trees used to analyze safety systems failures will be based on WASH-1400 experience and therefore will be quite comprehensive. Structural failure will be treated as just an additional mode of component failure. Identifiers on the components to locate them in the structure will aid if selective building failure is deemed as a probable mode of failure.

SSMRP is an ambitious project and it is hoped that by demonstrating methodology in phase 1 future research directions will become clear. Certain simplification in the treatment of human factors, design errors, and dependencies will be necessary in Phase 1. However, it is the intent to come up with a realistic analysis and the simplifications will result in wider ranges of uncertainties than would otherwise be the case. Since it is not clear how sensitive input variables will be to various outputs, a number of outputs are going to be calculated and studied. By studying these outputs it is expected that program objectives can be met.

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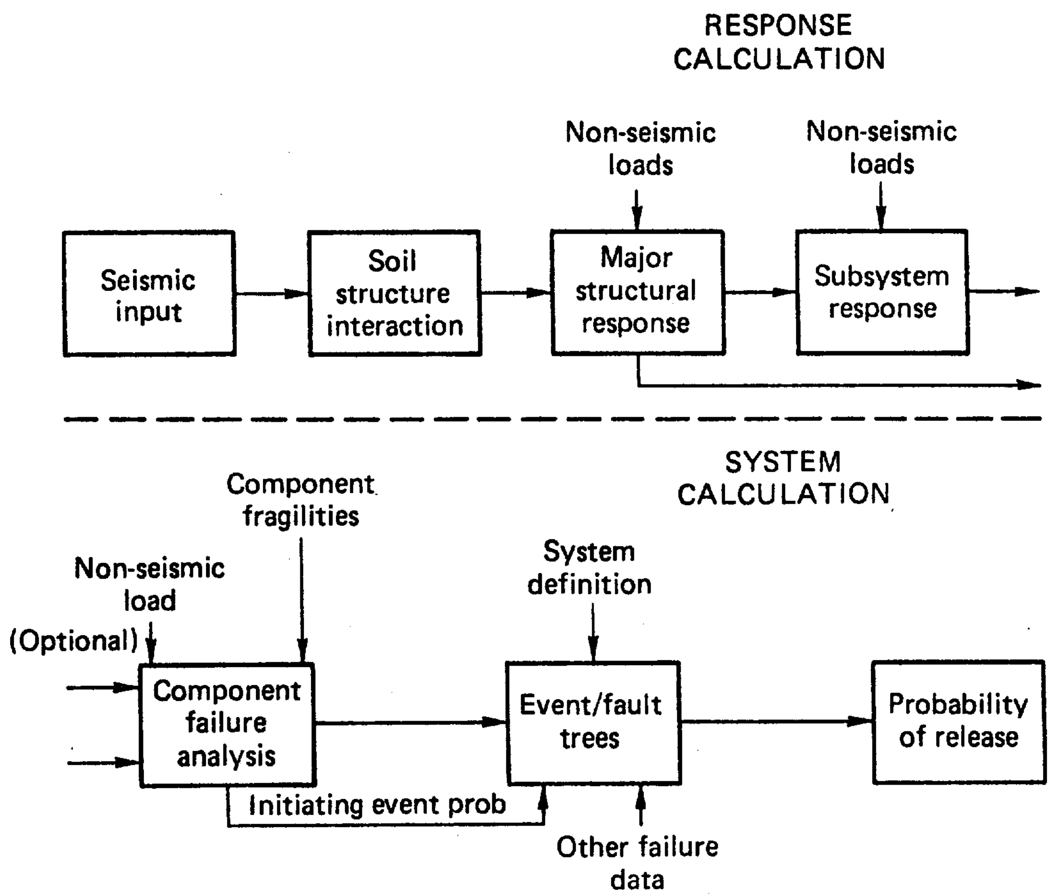
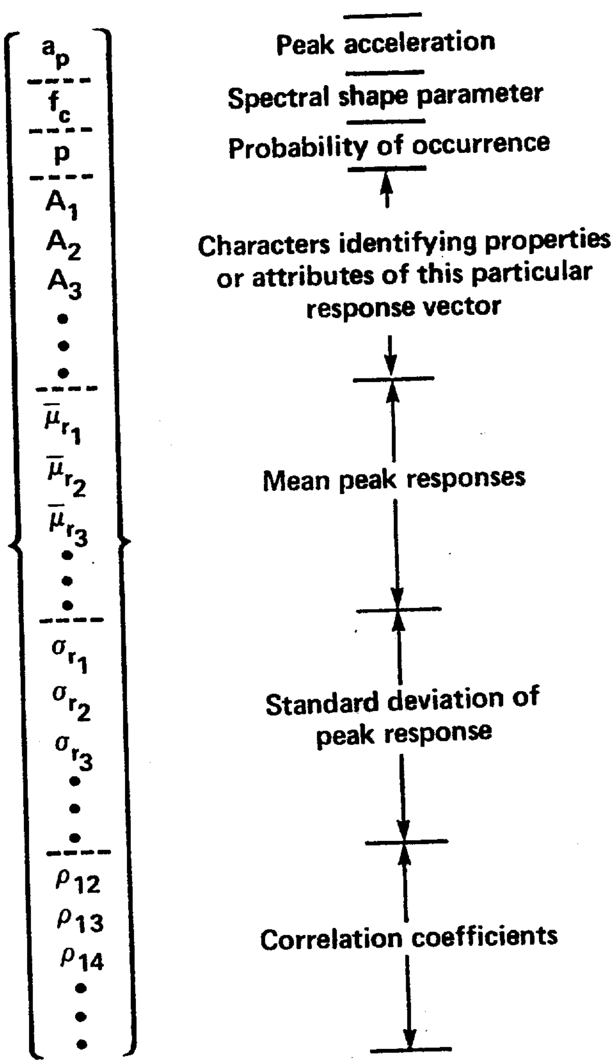
Table I. COMPARISON OF COMPUTATION OPTIONS

	Option number						
	1	2	3	4	5	6	7
Other than normal/lognormal distributions					•	•	•
Normal/lognormal distributions	•	•	•	•			
Correlation between component failure probability (due to response correlation)	•	•	•	•	•	•	
Analytical solution of failure probabilities	•						•
Monte Carlo solution of failure probabilities		•	•	•	•	•	
Covariance representation of fragility	•						
Fragility distribution function input (no correlation between fragilities)		•	•	•	•	•	•
Analytical solution of core melt sequence	•	•					•
Monte Carlo solution of core melt sequence:							
Probability form			•		•		
Binary form				•		•	

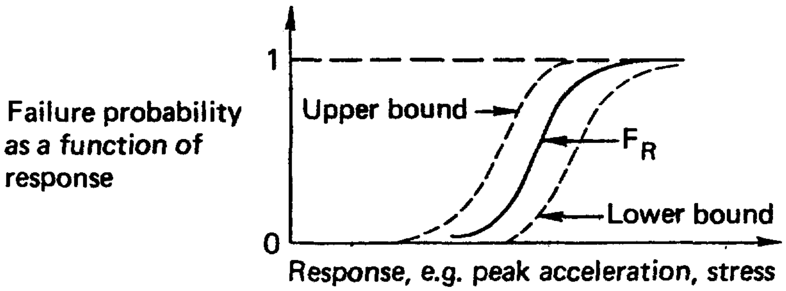
FIGURE CAPTIONS



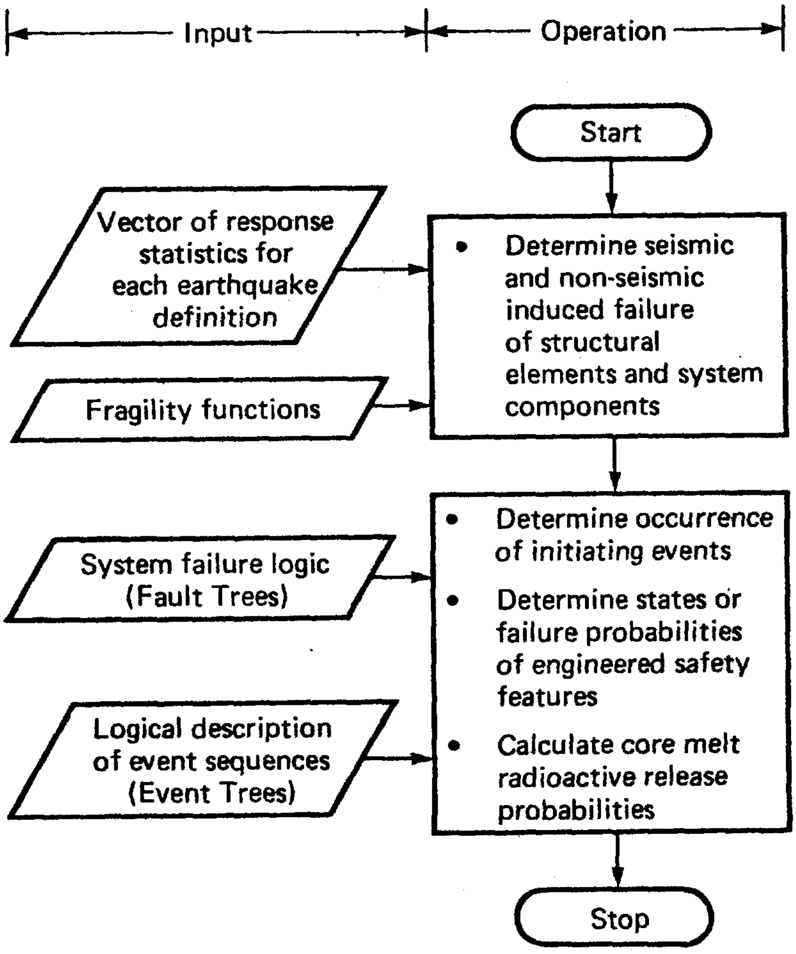
Cummings - Fig. 1



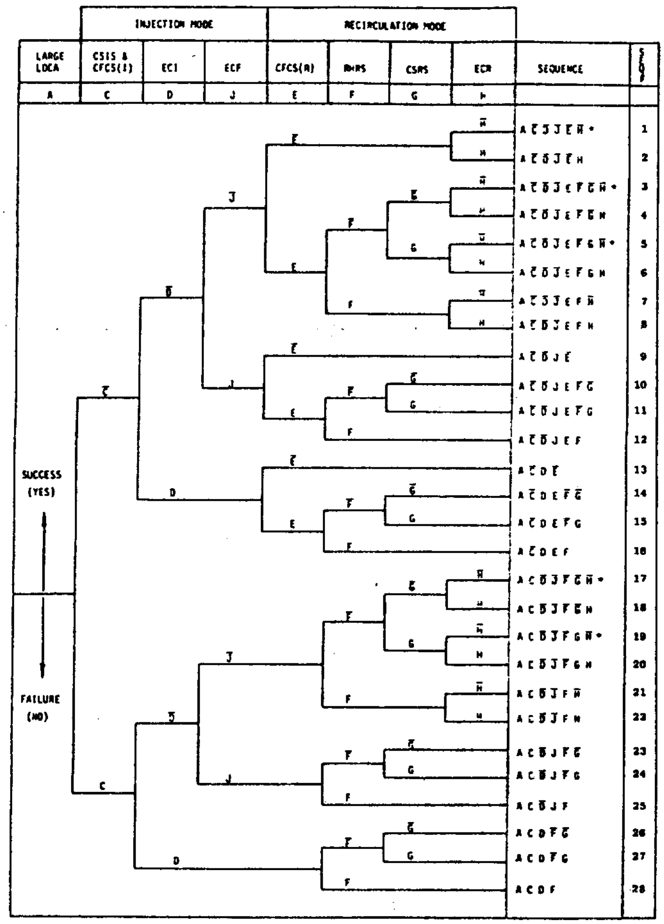
Cummings - Fig. 2



Cummings - Fig. 3

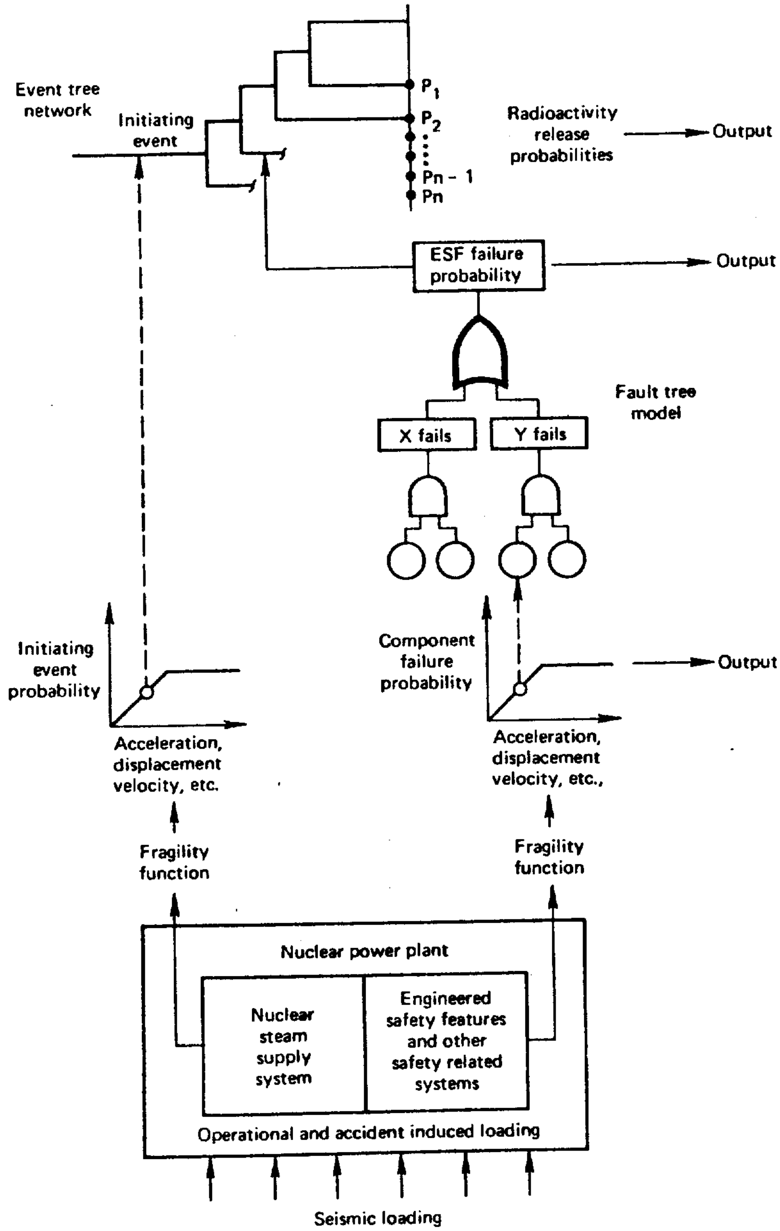


Cummings - Fig. 4



*NO CORE MELT

Cummings - Fig. 5



Cummings - Fig. 6